January 18, 2002

Dr. George E. Apostolakis, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

SUBJECT: NUCLEAR REGULATORY COMMISSION (NRC) ACTION PLAN TO ADDRESS THE DIFFERING PROFESSIONAL OPINION ISSUES ON STEAM GENERATOR TUBE INTEGRITY

Dear Dr. Apostolakis:

During the 486th meeting of the Advisory Committee on Reactor Safeguards (ACRS), October 4-6, 2001, the Committee reviewed the Action Plan developed by the Nuclear Regulatory Commission (NRC) staff to address the differing professional opinion (DPO) issues on steam generator tube integrity. The ACRS Subcommittee on Materials and Metallurgy had reviewed this Action Plan during its meeting on September 26, 2001. The conclusion of your review that the action plan appropriately and adequately responds to ACRS' recommendations concerning the DPO on steam generator tube integrity, was provided in a letter to the Commission dated October 18, 2001. This letter also provided detailed comments from the ACRS review on certain elements of the Action Plan. The enclosure summarizes the NRC staff response to the ACRS comments.

Sincerely,

/RA/

William D. Travers
Executive Director for Operations

Enclosure: As stated

cc: Chairman Meserve
Commissioner Dicus
Commissioner Diaz
Commissioner McGaffigan
Commissioner Merrifield
SECY

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DATE	12/10/01	12/10/01	12/10/01	11/30/01	11/30/01	12/7/01
OFFICE	NRR/DE*	NRR/DSSA*	NRR/ADPT*	TECH ED*	NRR/OD*	EDO
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STAFF RESPONSE TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) COMMENTS ON NUCLEAR REGULATORY COMMISSION (NRC) ACTION PLAN TO ADDRESS THE DIFFERING PROFESSIONAL OPINION ON STEAM GENERATOR TUBE INTEGRITY

ACRS Comment # 1

The efforts to understand threats to tube integrity posed by depressurization during main steamline breaks depend heavily on computer code analyses. In the absence of defensible, conservative load predictions, there is a need to validate prediction of computer codes with experimental data on modes of motion of steam generator tube support plates and stresses these motions place on steam generator tubes. As noted in NUREG-1740, extant experimental data on thermal hydraulics and forces on tube support plates during depressurization are suspect because of poor scaling of the experimental facilities.

Staff response

We are currently examining vendor analysis methods and experimental data used to validate codes for subcooled blowdown loads to help develop our analysis approach. Since a limited amount of adequately scaled data exist to validate computational tools for this application, we will generate conservative hydraulic forcing functions from first principles and use them as boundary conditions in the preliminary structural analysis. With respect to the thermal-hydraulic analysis, conservatism will be ensured by effectively neglecting the damping of the pressure wave by secondary side structures. This will maximize the amplitude of the forcing function imposed by the pressure gradient across the support plate and the tubes from the primary to the secondary side. A conservative structural analysis of the support plate and tube response will be performed by neglecting the damping imposed by the system structures and by applying the most conservative frequency of the forcing function. If adequate margin is not demonstrated with the conservative analysis, then we will perform a more refined analysis with a computational tool and will either validate it with existing experimental data or perform experimental testing as required.

ACRS Comment # 2

The NRC staff should actively participate in formulating and conducting the ARTIST tests to investigate decontamination on the secondary side of steam generators rather than simply waiting for the data from the tests to become available. Activities necessary to use and understand the data from the planned tests should be defined and included in the Action Plan.

Staff response

We are in the process of formally joining the ARTIST project and will actively participate in formulating the test matrix. We will also participate in pre-test and post-test analysis. Since the

Action Plan is of limited length and must cover the programs planned for each discipline, it provides an overview rather than the details of each activity. We will present the details of our plans to use the ARTIST data at subsequent ACRS meetings as requested.

ACRS Comment # 3

Plans for examining steam generator tube behavior under severe accident conditions are quite detailed. These plans should be augmented to include a detailed assessment of the understanding of loop-seal clearing and the subsequent behavior in the reactor coolant system.

Staff response

With respect to loop seal clearing phenomena, we agree with the ACRS. We have previously identified the importance of loop seal clearing on severe accident thermal hydraulics. We are continuing to address this behavior as a key part of ongoing analysis and will describe that detailed assessment in future discussions with the ACRS.

ACRS Comment # 4

We are impressed by the progress that has been made in the modeling of mixing and flow in the steam generator input plenum using CFD models. We believe that this work will serve as a good example of how the NRC can use CFD models to resolve complicated regulatory issues.

Staff response

The ACRS support of the computational fluid dynamics (CFD) modeling work is appreciated. We plan to continue the CFD application beyond the 1/7th scale steam generator assessment. Full scale steam generator modeling is currently underway.

ACRS Comment # 5

The lack of a correlation between leakage and voltage for 7/8-inch tubes is perplexing, in view of the good correlation for the 3/4-inch tubes. The staff should investigate the reason for this.

Staff response

In the voltage-based tube repair criteria described in Generic Letter (GL) 95-05, “Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking,” August 3, 1995, an empirical relationship is developed between the signal amplitude from a flaw (i.e., the bobbin coil voltage) and the amount of leakage from that flaw under postulated steamline break conditions. Separate correlations between bobbin voltage and leakage exist for 3/4-inch and 7/8-inch diameter tubes. The voltages used in these correlations are the voltages measured in the field prior to tube pulling.

Prior to issuing GL 95-05, the staff, with the aid of a contractor, performed statistical evaluations of the leakage and burst correlations. The staff evaluated whether the data from the two data sets (i.e., 7/8-inch and 3/4-inch) could be merged. These evaluations indicated that separate correlations for the 3/4-inch and 7/8-inch tubes were appropriate. In addition, the staff evaluated the statistical approach for evaluating the data and concluded it was acceptable and appropriately treated uncertainties in the correlation.

Both the 7/8-inch and 3/4-inch correlation exhibit scatter. Scatter in the data is not unexpected since the amount of leakage is a function of crack morphology; and the morphology of the crack networks at the tube support plates are very complex ranging from a small number of cracks to hundreds of cracks at tube support plate elevations. All of the available data are included in the correlations.

As the ACRS points out, there is more scatter in the 7/8-inch database (29 data points: 5 domestic pulled tubes, 2 foreign pulled tubes, 22 laboratory specimens), than the 3/4-inch database (48 data points: 14 domestic pulled tubes, 11 foreign pulled tubes, 23 laboratory specimens). Because of the range of variables affecting the leakage and voltage, the “limited” number of data points, and the availability and current state of the specimens, a simple explanation for the differences in the correlations could not be established. Nonetheless, we believe that the methodology used in predicting leak rates yields a conservative estimate of the leakage for the following reasons:

- Pre-pull voltages are used in the correlations. If the crack tears as a result of the tube pull operation, the measured leakage is expected to be higher than if the tube were not damaged (i.e., there is more leakage for a given voltage).

- The dose consequences are determined from the leakage evaluated at the 95th percentile at 95 percent confidence (i.e., a conservative value rather than a mean value is used).

- If the crack is plugged by deposits as a result of the test, the leakage would be reduced introducing more scatter in the data, which usually results in more conservative predictions (at the 95/95 level).

- If a statistical correlation cannot be demonstrated, the GL indicates that leakage should be treated as independent of voltage, which is conservative (since most indications left in service are “low” voltage indications, which tend to leak less than the mean, assuming leakage does increase with voltage).

In summary, we agree that the correlation between leakage and bobbin voltage is weaker for 7/8-inch diameter tubes than for 3/4-inch diameter tubes. As additional tubes are pulled in accordance with GL 95-05, the staff will continue to evaluate the database to determine if a correlation exists. This additional data may result in a stronger correlation for 7/8-inch tubes (or a weaker correlation for 3/4-inch tubes). Alternatively, additional 7/8-inch data may result in a weaker or even no correlation. A variety of parameters could affect the leakage and/or voltage data in the database; however, the data (as described above) do not lead to a simple conclusive explanation for the differences in the database. Finally, we believe the overall methodology for assessing the consequences of primary-to-secondary leakage is conservative. As a result of the above, we are not planning any additional efforts on this matter except to evaluate the effect of future pulled-tube data on the correlations.

ACRS Comment # 6

The proposed work in connection with developing a better understanding of radioactive iodine behavior under design-basis accident conditions suggest that the staff does not accept our recommendation. Certainly, the staff has not committed to develop further the existing, mechanistic models of the iodine spiking phenomenon.

Staff response

Steam Generator Action Plan (SGAP) milestone 3.9 was developed in response to the ACRS comments on iodine spiking contained in NUREG-1470, “Voltage-Based Alternative Repair Criteria." The milestone was to develop a more technically defensible position on the treatment of radionuclide release to be used in the safety analyses of design-basis events. We have performed statistical assessments of the existing data. Based on these assessments, we are developing a proposed response to the ACRS comments and recommendations presented in NUREG-1470, and will publish the response in the Federal Register for stakeholder comment.

Stakeholder comments should provide additional insights to help us determine the adequacy of the response and whether additional actions need to be taken, including the possibility of further developing mechanistic models of the iodine spiking phenomenon.

ACRS Comment # 7

The effort to develop a mechanistic understanding of stress corrosion cracking and its relationship to voltage signals, is very long-term in nature as would be expected. This work will be conducted under a continuing cooperative international research program on steam generator tube integrity.

Staff response

As indicated in the ACRS letter, the effort to gain a better understanding of stress corrosion cracking and its relationship to voltage signals is long-term in nature. However, insights related to crack initiation, evolution, and growth will be gained throughout the 5 year duration of NRC’s Third International Steam Generator Research Program to begin in January 2002. Our international partners in this program will contribute some of their own research results which will lead to a better understanding of stress corrosion cracking.